ANL-6692
Reactor Technology
(TID-4500, 19th Ed.)
AEC Research and
Development Report

ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois

A RETROSPECTIVE ANALYSIS OF ASPECTS OF THE ALPR (SL-1) DESIGN

by

W. J. Kann and D. H. Shaftman

Reactor Engineering Division

November 1962

Operated by The University of Chicago
under
Contract W-31-109-eng-38
with the
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ACKNOWLEDGMENTS

The authors wish to acknowledge the following Argonne National Laboratory personnel:

Mr. E. E. Hamer, for general comments and for discussions with regard to the development of the ALPR design.

Messrs. J. E. Draley and S. Greenberg, for discussions about corrosion of burnable-poison strips in hot water.

Messrs. R. A. Noland and D. E. Walker, for consultations with regard to the use of burnable poisons in reactors.

Messrs. C. R. Breden and N. R. Grant, for discussions concerning the use of soluble poison for control of boiling water reactors.

Messrs. R. O. Brittan, H. H. Hooker, W. C. Lipinski, G. C. Milak, A. Smaardyk, and B. I. Spinrad for consultations and numerous helpful suggestions.

A RETROSPECTIVE ANALYSIS OF ASPECTS OF THE ALPR (SL-1)* DESIGN

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I. INTRODUCTION

The Argonne Low Power Reactor (ALPR) was designed by Argonne National Laboratory (ANL) as a prototype of a nuclear power plant intended for DEW Line sites. It was constructed at the National Reactor Testing Station, Idaho, and was tested and operated briefly by ANL. In February, 1959, responsibility for operation of the plant (designated SL-1) was transferred to Combustion Engineering, Inc., as operating contractor. On January 3, 1961, the reactor was destroyed in a nuclear incident.

A primary purpose of this report is to examine various aspects of the control system of the ALPR (SL-1), in the perspective of the occurrence of the incident and of the investigations and discussions stimulated by the incident.

Another major purpose is to present ideas, fostered by these discussions, which are relevant to the evaluation of the safety of reactor design and operation, and, in particular, of control system design.

II. CHRONOLOGY OF DESIGN AND OPERATION

Primary purposes of this report are:

(1) to present a critique of aspects of the reactor design, in the retrospect of the operational history of the reactor, and the fact that the incident occurred; and

(2) to discuss certain concepts of the philosophy of reactor control.

*Throughout this report, the designation ALPR is used when reference is made to the Argonne Low Power Reactor as designed, tested, and operated by Argonne National Laboratory. The reactor designation SL-1 is used in discussions concerning the subsequent period of operation of the plant.

The critique of the design proceeds on two levels. On the one hand, if the reactor had been designed for purposes of experimentation, or for use as a civilian power plant, the plant design would have been different in a number of respects. However, one must keep in mind that the reactor was intended for certain, specific purposes which strongly influenced the reactor design. Therefore, it is useful to review here some of the goals of and criteria for the reactor design.(1) For convenience, the specifications for the reactor plant are set forth in Appendix A: "Technical Characteristics - Argonne Low Power Reactor (ALPR)."

The ALPR was designed as the prototype for a nuclear power plant which would meet the needs of an Auxiliary Station, DEW Line, and would be compatible with arctic environmental conditions.(2) In accordance with the specifications, the ALPR was constructed at the National Reactor Testing Station, Idaho, in an area set aside for military reactors.

As a prototype plant, the ALPR was to contain essentially only those extra features which were required for the safety of the personnel and of the plant, e.g., extra nuclear instrumentation used in the preoperational program of reactor physics experiments. In proving the design, it was intended that the reactor would be tested primarily for its capacity to meet the requirements of successful routine operation. Thus, it was expected that the plant would be in essentially sustained operation for a period of approximately three years. Training of military personnel was to be a part of the operation, but it was expected that the training would be subordinate to and coincidental with the routine operation of the plant.(3)

In further accordance with the design specifications, the reactor was designed as a thermal, heterogeneous system, fueled with assemblies of clad plates containing uranium highly enriched in U^{235}, and cooled and moderated by the natural circulation of water within the pressure vessel containing the core. The normal means for controlling the reactor was a combination of control rods, containing natural cadmium, and a set of burnable-poison strips (highly enriched boron, dispersed in aluminum-nickel) which were fusion-welded to side plates of fuel assemblies. In addition, the design included a backup, non-emergency control system based on the introduction of a solution of boric acid into the reactor water. (A more detailed description of plant components is given in Section IV.)

As designed and built, the reactor core structure provided spaces for as many as 59 fuel assemblies, plus one neutron-source assembly. The reference ALPR reactor, with a nominal thermal power rating of 3 Mwt, included 40 fuel assemblies and five cross-shaped control rods. However, additional spaces, for four tee-shaped control rods, were available at the periphery of the full core structure.

Critical experimentation began on August 11, 1958, at which time a critical array, consisting of ten fuel assemblies without poison strips, was attained at room temperature. During the ensuing weeks, an experimental program was carried out both at room temperature and at operating temperature (~420°F), and a final reference core was selected for use in the prototype plant. Details of the zero-power experimental work are given in Ref. 4. A chronology, both of these pre-power experiments (Experiments No. 1 to 122), and of the subsequent tests at power (Experiments No. 123 to 210B), is supplied in Ref. 5.

On October 23, 1958, a series of check criticals were begun at room temperature, and were continued to various temperatures up to the design operating value, 420°F. In this sequence, nuclear power (~$\frac{1}{2}$ Mwt) was used to heat the reactor water. In continuation of this work, the reactor power level was raised, in steps of 1,000 lb steam/hr, to a steam mass flow rate of 9,150 lb/hr, at which time (October 24, 1958) the turbine-generator equipment was placed in operation.

Between October 24, 1958 and November 10, 1958, power runs were made to determine such reactor characteristics as:

(1) the reactivity effect of equilibrium xenon at full power;

(2) the buildup of maximum xenon after reactor shutdown from full power;

(3) relative positions of the five cross-shaped control rods as a function of the measured steam mass flow rate;

(4) response of the reactor to "step-function" changes in feedwater flow rate, and in the mass flow rate of steam to the turbine;

(5) the proper functioning of steam safety and pressure-relief valves;

(6) response of the reactor, when under automatic power demand control, to programmed changes in power demand; in these runs, various settings of the dead band and the travel ratio on the automatically operated drive for the center control rod were tested;

(7) the proper operation of the safety devices which shut down the reactor when positions of high or low water level (in the vessel) are reached; and

(8) the response of the reactor to "blind startup."

Also, during this period, tests were made to investigate the response of the turbine-generator equipment to programmed sequences of step changes in load demand.

Between November 19, 1958 and December 11, 1958, a 500-hr plant performance acceptance test was completed, with a chargeable down-time percentage of less than 3%. During this time, the automatic power demand system was modified, temporarily, to investigate the response of the reactor to a control system in which the position of the center control rod was regulated by a signal of the deviation of reactor pressure from a fixed value.(5)

The period between December 12, 1958 and February 5, 1959 was devoted to:

(1) additional training of military personnel in the routine operation of the plant;

(2) modifications of the plant to correct various deficiencies observed;(6)

(3) additional plant performance tests (January 5, 1959 to January 9, 1959);(6)

(4) a permanent modification of the automatic control system, as described in the preceding paragraph, and power operation which verified that the new controller was satisfactory; and

(5) a period of training of personnel of the new operating contractor, Combustion Engineering, Inc.

On February 5, 1959, Combustion Engineering, Inc. became the operating contractor of the SL-1.

Details of the subsequent history of reactor operations are available in a number of reports by Combustion Engineering, Inc. (C.E.), some of which are listed in the References of Ref. 7. In Section VII of the present report, some of the problems noted by C.E. during their period of reactor operation are described.

On January 3, 1961, after approximately ten days of end-of-year shutdown, and during the process of reassembling the drive mechanisms to their control rods, a nuclear incident occurred resulting in the destruction of the reactor core and the death of the three military personnel on duty.

III. THE SL-1 INCIDENT

The men on duty at the time of the incident had been assigned a series of tasks to perform during that shift period of eight hours. It appears that, at the moment of the incident, all three men were standing on the top head of the pressure vessel. The official reconstruction of their activities is as follows:(8)

"At the time of the explosion, the reactor crew was most probably engaged in reassembly of the number 9 control rod (the central control rod) assembly. To the Board, it appears quite plausible that the shift supervisor and the other regular member of the crew were located on top of the reactor vessel at the time of the explosion. The third member of the crew, who was a trainee, might have been partially over the reactor top or close to the edge of the reactor top. Based on medical evidence of injury, location of the men after the incident and knowledge of the individual capabilities and job assignments, it appears reasonable to hypothesize that the supervisor was in a crouched or squatting position which would be normal for manipulation of the C-clamp during the reassembly of the number 9 control rod drive. Similarly, it would appear reasonable to hypothesize that the regular crew member was located in a position for lifting the number 9 control rod assembly with the handling tool. There is no direct evidence to corroborate these hypothetical assertions."

In their final report,(9) summarizing the evidence obtained during their 13 months of cleanup and investigation of the SL-1, the General Electric Co. (G.E.) group outlined a possible sequence of events, following a manual withdrawal of the center control rod. In this postulated sequence, shown in Fig. 1, a minimum reactor period of approximately 4 ms is derived, corresponding to a maximum reactivity insertion of approximately 2.4%.

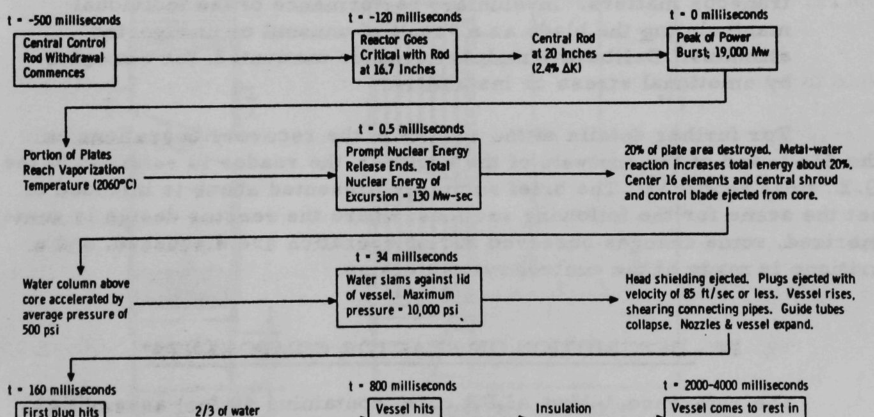

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Fig. 1. SL-1 Excursion Summary (Reproduced from Fig. IV-4 of Ref. 9. Values are approximate as discussed in Ref. 9.)

Although the accuracy of some of the details included in Fig. 1 is open to question, there is little doubt that the incident was caused by a manual withdrawal of the center control rod far beyond the maximum position specified in the instructions for assembling the drive mechanism to its control rod.[9]

Attempts to pinpoint the reason for a rod withdrawal apparently so much in violation of the specified assembly procedure have been unsuccessful. The extensive history of earlier assembly operations of this type, for the center control rod, offers no indication of difficulty. On the morning of January 3, 1961, a disassembly of the center rod from its drive was accomplished without difficulty. In their post-incident work, G. E. found no evidence which might indicate that the center rod had stuck and that an attempt had been made to pull it loose. (On the other hand, there was no proof that this was not the cause of the incident.)[10]

In the letter of submission of their final report, the Board of Investigation for the SL-1 Incident stated:[11]

> "One or more of the following circumstances may have resulted in the abnormal manual withdrawal of the blade:
>
> Inadequate or faulty procedure or training. (The Board has no reason to change its previous conclusion that the training of the military personnel for this maintenance operation was adequate.) 'Human error,' for example, incorrect manipulation of the blade owing to preoccupation of the manipulator with extraneous matters. Involuntary performance of the individual manipulating the blade as a result of unusual or unexpected stimulus. Deliberate malperformance motivated, for example, by emotional stress or instability."

For further details of the results of the recovery operations on the SL-1, and of the analysis of the accident, the reader is referred to the G. E. final report.[9] The brief summary presented above is intended to set the scene for the following sections, where the reactor design is summarized, some changes observed during operation are discussed, and a critique is made of the control system design.

IV. DESCRIPTION OF REACTOR COMPONENTS*

The reference 3-Mwt ALPR core, containing 40 fuel assemblies, was controlled by a system of five cross-shaped control rods, and full-length and half-length burnable-poison strips attached to the side plates

*A more detailed description of components can be found in Ref. 1.

of fuel assemblies. The core shroud structure provided spaces for as many as 59 fuel assemblies, plus one neutron source assembly. For control of so large a reactor, channels for four tee-shaped control rods were available. In the reference core, dummy fuel assemblies were inserted so as to fill compartments and, thereby, to keep the fuel assemblies and the source assembly in vertical alignment. A manually operated boric acid system was available for slow, non-emergency shutdown.

A. Fuel Assembly

The reference core contained 40 fuel assemblies. Each fuel assembly (Fig. 2) contained nine fuel plates. The active portion of the fuel plate consisted of a center zone (0.050 in. x 3.50 in. x 25.8 in. long), of an

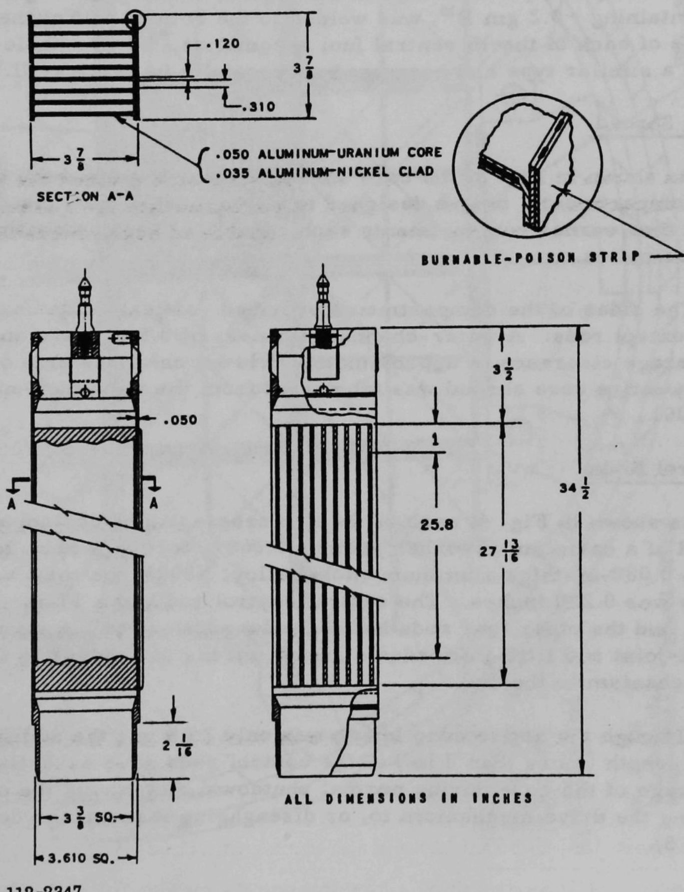

Fig. 2

Fuel Assembly

alloy of uranium, aluminum, and nickel. The uranium was enriched to ~93 wt-% U^{235}; the nickel constituted ~1.6 wt-% of the fuel-plate core.(12) The fuel-plate core was clad on both faces with a 0.035-in. thickness of aluminum-nickel alloy, X8001.

B. Burnable-Poison Strips

Thin burnable-poison strips were fusion-welded to one or both side plates of the fuel assembly. The strips, extruded from a mixture of aluminum-nickel and boron highly enriched (>90%) in B^{10} (boron-10), assisted in the control of the excess reactivity of the core. In the reference core, one full-length (25.8 in.) poison strip, containing ~0.5 gm B^{10}, was welded to a side plate of each fuel assembly. One half-length (12.9 in.) strip, containing ~0.2 gm B^{10}, was welded to the bottom half of the other side plate of each of the 16 central fuel assemblies.(13) Burnable-poison strips of a similar type had been used successfully in BORAX-III.(14)

C. Core Shroud

As shown in Fig. 3, the core shroud structure divided the core into 16 compartments, twelve designed to accommodate four assemblies each and four corner compartments each capable of accommodating three assemblies.

The sides of the compartments provided vertical guide channels for the control rods. A water-channel thickness of 0.5 in. gave an adequate average clearance of approximately 0.14 in. on either side of the rod. The entire core shroud was fabricated from the aluminum-nickel alloy X8001.

D. Control Rods

As shown in Fig. 4, each of the five cross-shaped control rods consisted of a cadmium absorber section (0.060 in. x 14 in. x 34 in. long) clad with 0.080-in.-thick aluminum-nickel alloy, X8001; the total rod thickness was 0.220 inches. The central control rod had a 17-in.-long follower, and the other four rods had 5-in.-long followers. A stainless steel ball-joint end fitting afforded a simple means of connecting the drive mechanism to the rod.

Although the active core length was only 25.8 in., the additional absorber length (more than 8 in.) of the control rods gave essentially full coverage of the core during normal shutdown, and during the process of engaging the drive mechanism to, or disengaging it from, the control rod (Fig. 5).

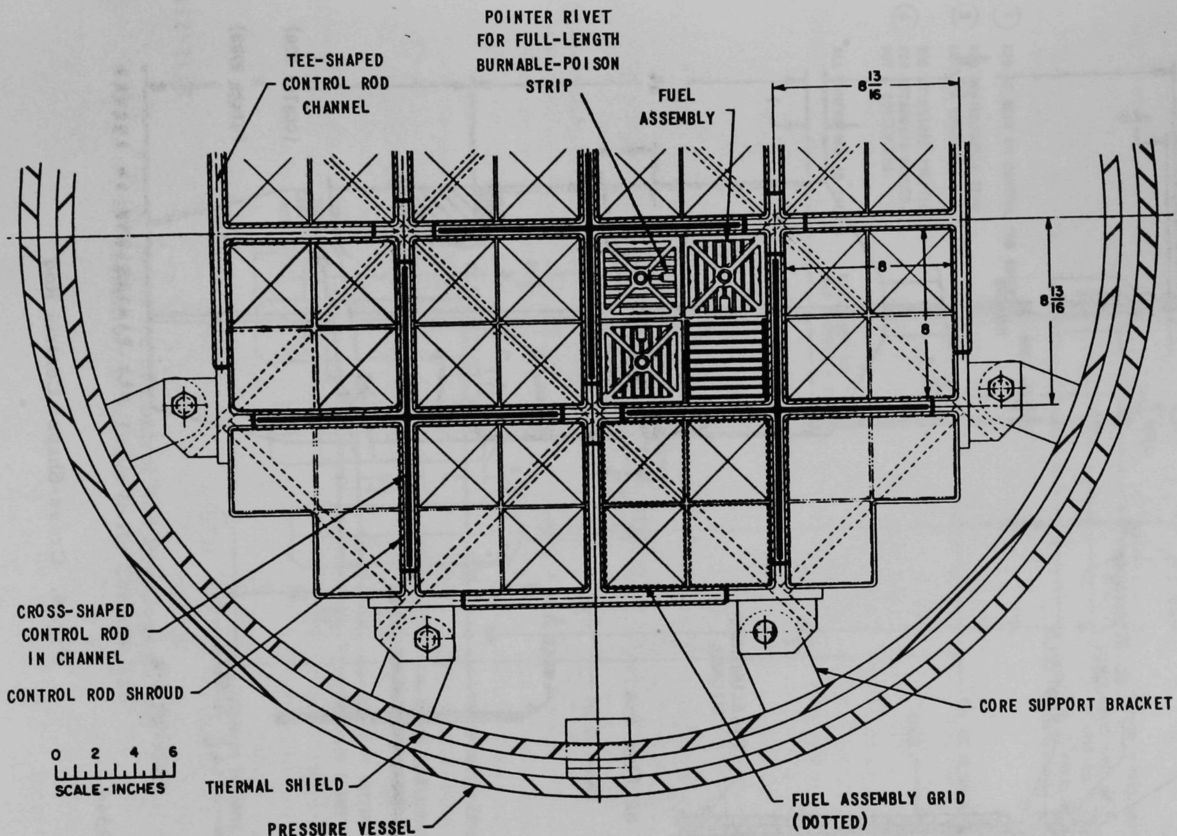

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Fig. 3. Core Plan, Section

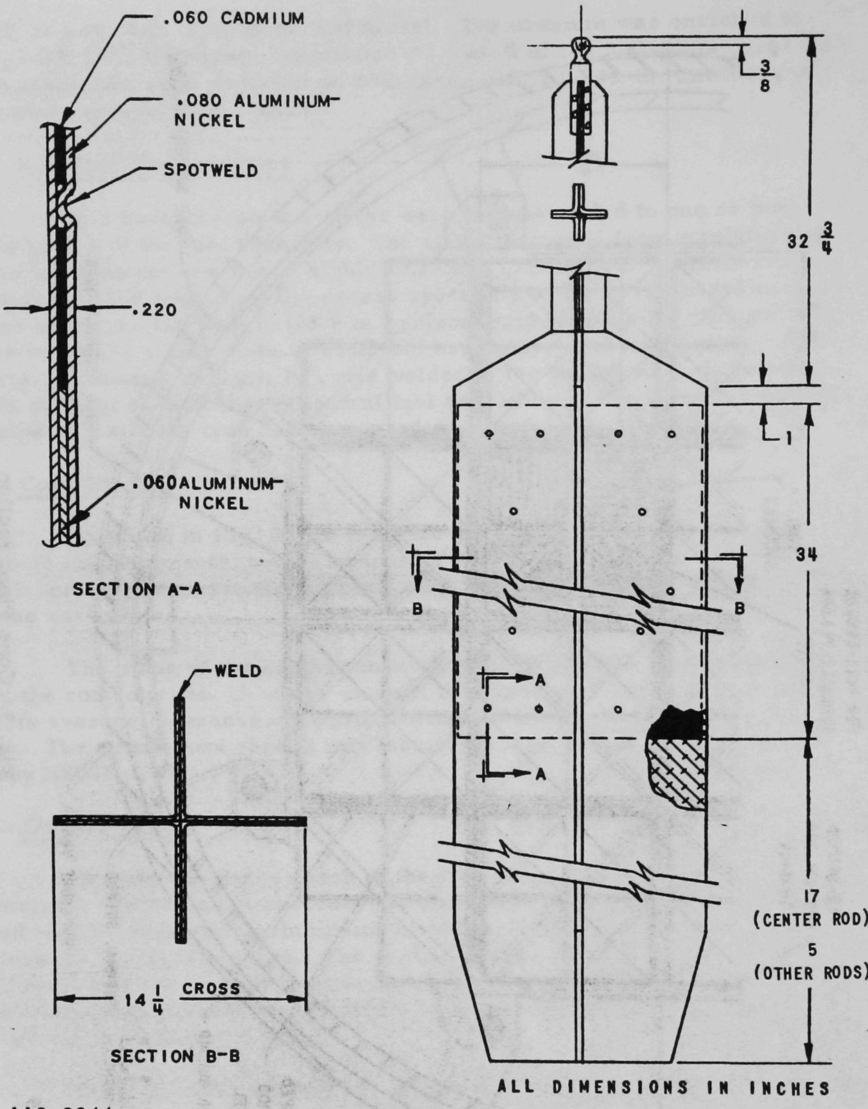

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Fig. 4. Cross-Shaped Control Rod

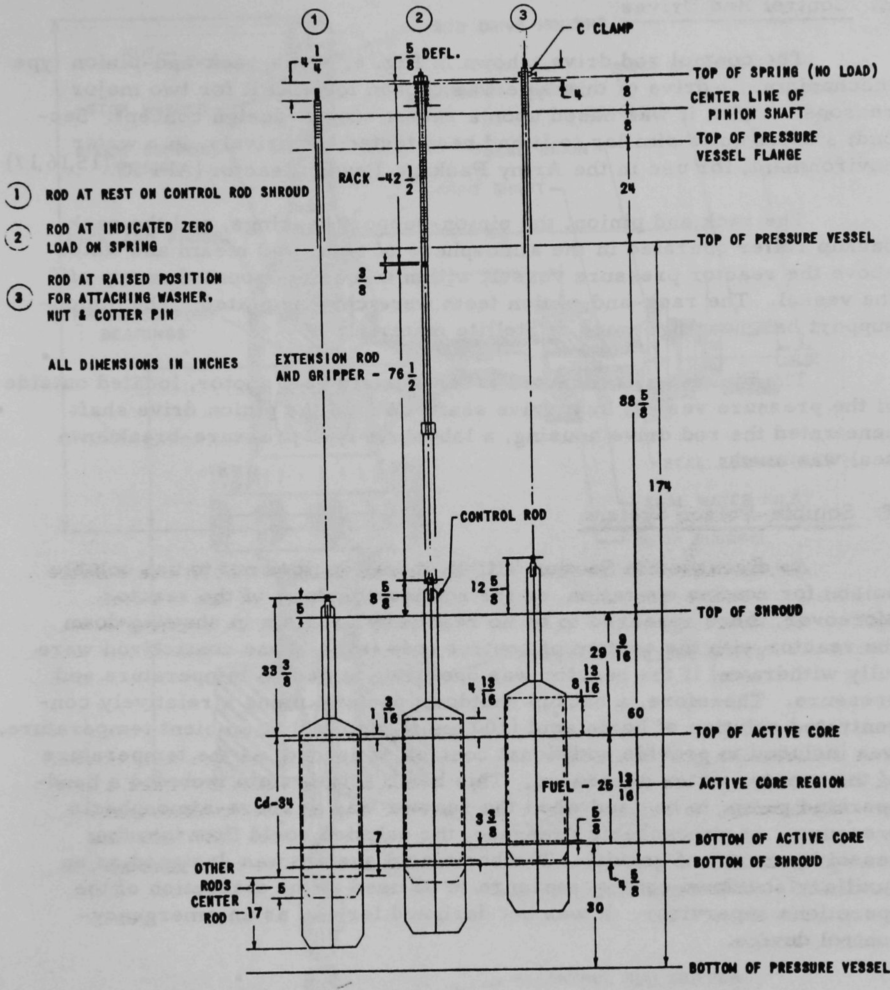

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Fig. 5. Sequence of Control Rod Positions during Assembly of a Drive to Its Control Rod

E. Control Rod Drives

The control rod drive, shown in Fig. 6, was a rack-and-pinion type mechanism. A drive of this type was chosen for ALPR for two major reasons. First, it was based upon a rather simple design concept. Second, a drive quite similar to it had been tested intensively, in a water environment, for use in the Army Package Power Reactor (APPR).[15,16,17]

The rack and pinion, the pinion-support bearings, and the rack backup roller operated in the atmosphere of saturated steam and water above the reactor pressure vessel, within a housing mounted on top of the vessel. The rack-and-pinion teeth were chrome plated. The pinion-support ball bearings were of Stellite material.

The pinion was connected to its electric gear motor, located outside of the pressure vessel, by a drive shaft. Where the pinion drive shaft penetrated the rod drive housing, a labyrinth-type pressure-breakdown seal was used.

F. Soluble-Poison System

As discussed in Section VIII-D, it was decided not to use soluble poison for routine operation, or for routine shutdown of the reactor. Moreover, there appeared to be no reactivity problem in shutting down the reactor with the system of control rods (even if one control rod were fully withdrawn) if the reactor was operating at design temperature and pressure. Therefore, a backup shutdown system, using a relatively concentrated solution of boric acid (100 gm H_3BO_3/gal) at ambient temperature, was included to provide additional control, if needed, as the temperature of the reactor water decreased. This boric acid system included a hand-operated pump, to be used when the reactor was at above-atmospheric pressure. At atmospheric pressure, the solution could flow into the vessel by means of gravity. The boric acid system was designed as an auxiliary shutdown control measure to be used at the discretion of the operations supervisor. It was not designed for use as an emergency-control device.

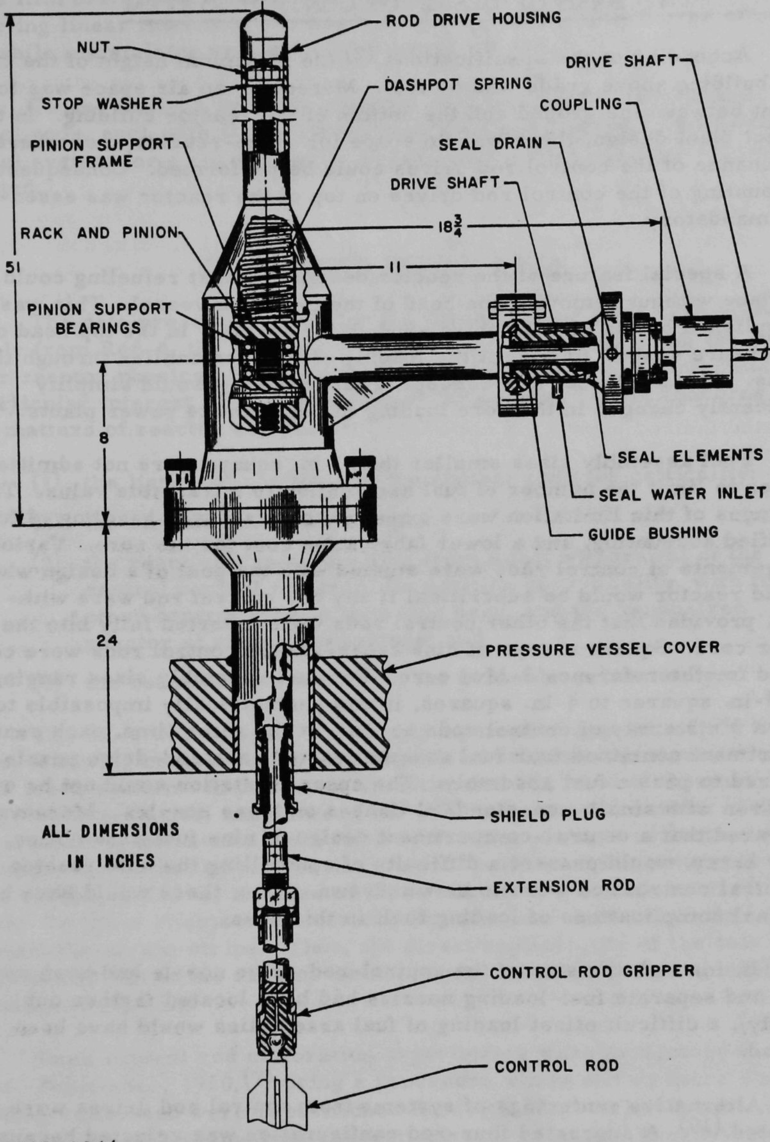

Fig. 6. Control Rod Drive

V. BASIS OF DESIGN OF CONTROL SYSTEM

According to the specifications,[2] the maximum height of the reactor building above grade was 50 feet. Moreover, an air space was to be present between the ground and the bottom of the reactor building. In this compact plant design, there was no space for a sub-reactor area where maintenance of the control rod drives could be performed. Consequently, the mounting of the control rod drives on top of the reactor was essentially mandatory.

A special feature of the reactor design was that refueling could take place without removing the head of the pressure vessel. This was accomplished by sizing the control-rod-drive nozzles, in the top head of the pressure vessel, to permit the loading of fuel assemblies through the nozzles. It was felt that this concept of fuel handling would simplify significantly changes in the core loading in the ultimate power plants.[18]

Fuel assembly sizes smaller than 3 in. square were not admitted in order to limit the number of fuel assemblies to a tractable value. The advantages of this limitation were expected to be reduced handling of fuel, simplified accounting, and a lower fabrication cost for the core. Various arrangements of control rods were studied with the goal of a design where the cold reactor would be subcritical if any one control rod were withdrawn, provided that the other control rods were inserted fully into the reactor core. Square arrays of nine cross-shaped control rods were considered for the reference 3-Mwt core. For fuel assembly sizes ranging from 3-in. squares to 4-in. squares, it was geometrically impossible to set up a 3 x 3 array of control rods so that, at the same time, each central compartment contained four fuel assemblies and each rod-drive nozzle was sized to pass a fuel assembly. The space limitation could not be overcome even with small, non-standard flanges on these nozzles. Moreover, it appeared that a central-compartment design of nine fuel assemblies, in a 3 x 3 array, would present a difficulty of controlling the cold reactor if the central control rod were to be withdrawn. Also, there would have been additional complications of loading fuel, in this case.

If, instead, the size of the control-rod-drive nozzle had been reduced, and separate fuel-loading nozzles had been located farther out (radially), a difficult offset loading of fuel assemblies would have been required.

Alternative centerings of systems from control rod drives were investigated.[19] A suggested four-rod configuration was rejected because of the difficulty of control with <u>any</u> one rod withdrawn. A design of nine control rods and five drives was examined. In this concept, each of four drives controlled a pair of control rods, and the center rod was controlled by the fifth drive. This design was rejected because of the problem of imparting linear motion to two rods, by a drive not centered over either rod, while maintaining vertical alignment of the rods.

Consequently, a reference design employing five cross-shaped control rods, each with its own drive mechanism, to serve as a movable-control system for a core of forty, $3\frac{7}{8}$-in.-square fuel assemblies, was selected.

VI. REACTOR PHYSICS DATA

The data presented below for the fresh reactor have been taken largely from Ref. 4, the Argonne National Laboratory report on the zero-power reactor physics experiments on the fresh ALPR. The information of particular interest for the purposes of the present report concerns such matters of reactor control as:

(1) the listing of positions of the control rods for criticality;

(2) the reactor shutdown margin;

(3) the reactivity gain resulting from a full withdrawal of the central control rod from its position of criticality, with the other control rods at indicated zero, and the xenon-free reactor at room temperature; and

(4) the reactivity controlled by the burnable-poison strips.

The SL-1 was in operation during a period of approximately two years, following completion of the program of critical experimentation on the fresh system. In that time, an estimated 931 Mwd of thermal energy were produced, and some of the U^{235} fuel atoms and much of the B^{10} burnable-poison atoms were destroyed in the process of neutron absorption. The values of the reactor-control parameters enumerated above varied with time, because of these effects of the production of power. To these effects one must add the influence of deterioration of the burnable-poison strips. Thus, the direct applicability of the data for the fresh reactor to the case of the reactor at the time of the incident is seriously in question.

Some control rod calibration experiments were performed in August-September, 1960,[7] using a procedure where one or more rods not being calibrated were repositioned for criticality. It has been noted that such a procedure involves corrections to account for interaction effects between control rods. Since the degree of loss of B^{10} due to strip deterioration is unknown, only the first-listed parameter (critical rod position) is known reasonably well as a function of the cumulative energy output of the reactor.

A. The Cold Fresh Reactor (no Xe^{135} or Sm^{149})(4)

In the fresh reference 3-Mwt reactor (ALPR Loading No. 57), criticality was achieved at 94°F with the center control rod at an indicated position of 19.1 in. and the other four cross-shaped control rods at indicated zero (fully inserted). Alternatively, the reactor was critical with the five cross-shaped control rods banked at an indicated position of ~12.6 inches.

At room temperature, this loading was subcritical with the center control rod plus any three other control rods fully inserted. Without poisoning by Sm^{149} or Xe^{135}, it is doubtful that shutdown of the cold system could have been accomplished with two off-center rods withdrawn. However, early in core life, but after Sm^{149} had accumulated, it was possible to keep the reactor shut down with two off-center rods at 30 inches. Probably the reactor could have been shut down throughout its operating life in the absence of one off-center control rod.

As has been noted (Section IV), the final, fresh reference system included 40 full-length poison strips, each containing (nominally) 0.5 gm B^{10}, and 16 half-length strips, each with ~0.2 gm B^{10}. In the course of the zero-power reactor physics experimentation, an initial 40-fuel-assembly core was studied where only the full-length strips were present. In this initial system, the reactivity shutdown margin was ~5 dollars (or ~$3\frac{1}{2}$%). Upon adding the half-length strips, the shutdown margin was made still larger, though not by much, since to some extent the half-length strips were shadowed by the adjacent control rods.

Two differential-worth measurements were made on the center control rod, with the other four rods at zero.* At an indicated position of 19.3 in., the differential reactivity worth was ~1.1 dollars/inch. At 22.4 in., the worth of the center rod (No. 9) was ~0.57 dollars/inch. In each case, the differential worth was roughly $1\frac{1}{2}$ times the differential reactivity worth of rod No. 9 as measured from the five-rod bank. Assuming that this factor ($1\frac{1}{2}$) applied over the range from 19 in. to 30 in., and using Fig. 32 of Ref. 4, it is estimated that the cold, fresh reactor would have been supercritical by $4\frac{1}{2}$ dollars with the center control rod at 30 inches.

The total reactivity worths of burnable-poison strips were measured with enough boric acid dissolved in the reactor water to attain a five-rod-bank position of 20 in. at criticality. The 40 full-length strips controlled ~13 dollars; the 16 half-length strips controlled ~5 dollars.

*Unpublished.

B. The Cold Operated Reactor (no Xe^{135})

The control rod calibrations performed in the fresh reactor, in the Fall of 1958, involved the addition of a soluble neutron poison, boric acid (H_3BO_3), to the reactor water so as to calibrate rods from various rod bank positions. In August-September, 1960, some control rod calibrations were made using a procedure of attaining criticality by the repositioning of one or more of the rods not being calibrated.[7] Moreover, the strong neutron source was left in place, and this may have introduced perturbations and uncertainties in the positive-period rod calibrations.

It was noted above that the half-length strips provided little extra reactivity control when the control rods were inserted fully, because these strips were positioned near the surfaces of the thermally-black control rod blades. However, as the adjacent control rod was withdrawn, each half-length strip became more effective. As a result, the curve of differential reactivity worth of the rod versus rod position, obtained for the initial reference core without the half-length strips, was shifted upward and was distorted when the half-length strips were added. Thus, one major effect of the removal of B^{10} from these half-length strips, as a result of neutron capture and strip deterioration, was to tend to shift the differential worth curve back to the curve realized in the fresh reactor in the absence of half-length strips. Another effect of power production was to reduce preferentially the U^{235} content of fuel assemblies in the central 16 fuel assemblies and in the bottom half of the reactor core. Two effects of this preferential burnup were:

(1) to reduce the over-all worth of the center control rod, because of the relatively lower fuel content near the center of the reactor core; and

(2) to increase the control capacity of this rod, because of the increase in the "thermal migration area" in the vicinity of the rod.

It is the opinion of the authors that the net effect of actual reactor operation was to increase the differential worth of the central control rod in the bottom half of the reactor core, relative to the differential worth curve obtained in the fresh reactor.[20] If this is indeed the case, then the curves of "estimated shutdown margin"[21] underestimate the per cent subcriticality in the operated reactor in the situation where all five control rods were inserted fully. The conclusion would be that even without the total of six extra cadmium blades added in two tee-rod locations in November, 1960 (Section VII-A), throughout its period of operation the cold, xenon-free reactor would have been subcritical by more than 3 dollars with all five cross-shaped control rods at indicated zero.

There is no measured datum point for the critical position of the center rod, with the other rods at zero, on the morning of the day of the incident. An estimate of 16.7 in. has been reported;[22] this point is shown on Fig. 7. Thus, there appears to have been a margin of at least one (1) ft between the point of delayed criticality and the maximum position specified in the written instructions[23] for assembling the control rod drive to its rod (Fig. 5).

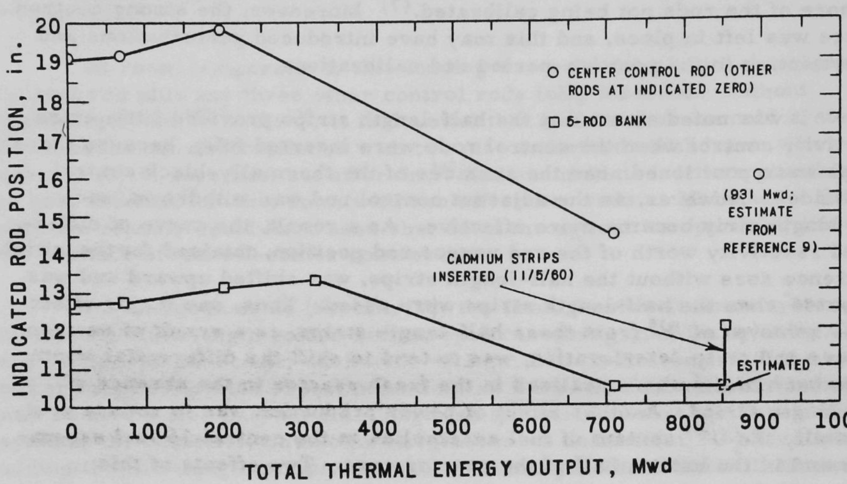

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Fig. 7. Critical Positions of Control Rods at Room Temperature vs Total Energy Output (No Xenon-135). [Data shown have been taken from Ref. 24]

VII. EVALUATION OF THE PERFORMANCE OF THE CONTROL SYSTEM

A. Burnable-Poison Strips

After approximately 400 Mwd of reactor operation (February, 1960) it became evident that the actual rod bank position, at power, was dropping much more rapidly than had been expected.

During a periodic inspection of fuel assemblies, in August, 1959, it had been observed that some bowing of burnable-poison strips had occurred between fusion welds which held the strips to the side plates of fuel assemblies.[25] In August, 1960, a second inspection of fuel assemblies revealed that poison strips were deteriorating.[25] Removal of fuel assemblies from the region of the center of the reactor core was accomplished only after considerable difficulty. These assemblies were bound tightly within the four-assembly compartment, probably because of the extreme bowing and deterioration of the poison strips in this region of higher power density. In the peripheral region of the reactor core, fuel assemblies were not held tightly, and the burnable-poison strips appeared to be in much better condition. (These remarks about the condition of burnable-poison strips are based on the remote viewing of a small number of samples at some distance under water.)(25)

It is probable that, in part, the progressive lowering of the critical position of the control-rod bank during reactor operation was due to a more rapid loss of B^{10} from the reactor core, because of burnup and/or strip deterioration, than had been anticipated. To compensate for this apparent loss in stationary reactivity control, in November, 1960, three cadmium blades were inserted in each of two diametrically opposite locations of tee-shaped rods.(26) These served as fixed, tee-shaped control rods in the reactor core.

From accelerated-corrosion experiments performed at Argonne National Laboratory, using water at 500°F, it appears that the type of bowing exhibited by the SL-1 poison strips would occur in the absence of neutron radiation, simply because of strip growth resulting from exposure of the constrained thin plates to hot water. The bowing reflects the forces of this volume expansion between the fusion welds which defined the regions of constraint.(27)

Personnel of Combustion Engineering, Inc., have reported results of various tests performed on samples of poison strips, including, among others:(28)

(1) metallographic inspection both of irradiated and of unirradiated samples;

(2) corrosion tests in a water environment approximating the operating conditions of SL-1 except for the absence of a neutron radiation field.

For the corrosion experiments, irradiated samples were taken from poison strips which had been in the SL-1 core. Unirradiated samples were taken from an unused poison strip.

Some summary remarks and conclusions of particular interest are:(28)

(1) "A dependency of corrosion rate on boron burn-up was found by post-irradiation corrosion tests. The section of one of the recovered poison strips taken from the high flux end of the

remnant exhibited a calculated penetration after 42 days of test greater than 10 times that of either an unirradiated sample or sections taken from lower flux regions of the same recovered poison strip. Corrosion behavior of the lower flux sections were similar to unirradiated samples. These data are indicative of a possible threshold burn-up value for accelerated corrosion attack."

(2) "Further evidence of the possible boron burn-up dependency of corrosion behavior was obtained from visual examination of entire lengths of poison strip remnants in which excessive pitting was observed to be concentrated near the irregular (high burn-up) end. Metallographic sections through these regions confirmed the existence of gross general corrosion in these areas."

(3) "Swelling of the Al-B plates on the order of 12% occurred as a result of the helium generated during the $B^{10} \xrightarrow{n,\alpha} Li^{7}$ transmutation."

(4) "There is a high probability that the failure of these poison strips was (sic) attributed to either excessive corrosion, corrosion induced embrittlement, or a combination of corrosion induced mechanisms together with porosity and crack formation occurring at the high burn-up regions of the poison strips."

B. Sticking of Control Rods

During reactor operation, a history of 84 occasions of "sticking of the control rods" was accumulated. This "sticking" has been categorized into three types:(29)

"Type I - Sticking of a control rod resulting in failure to meet the drop time requirements (one second for 10" drop; two seconds for a 30" drop) and which did not require a power assist from the drive assembly."

"Type II - Sticking of a control rod in which the control rod stopped and required a power assist to enable the control rod to reach its zero position (even if it subsequently fell free at a lower level)."

"Type III - Sticking of a control rod in which it was not possible to drive the control rod in a desired direction, e.g., clutch slippage during a rod withdrawal, or failure of a drive assembly shear key or gears resulting in failure to drive a control rod."

There were 19 Type-I, 52 Type-II, and 13 Type-III stickings.

So far as we know, no measurements were made, with the reactor at operating temperature and pressure, to check the dimensions of the pressure vessel head with regard to alignment of the drive mechanisms and the control rods. However, it is difficult to see why, if there had been misalignment of a control rod drive relative to the control rod guide shroud, stickings did not occur in a much greater percentage of the rod movements.

During the early post-incident investigations, it was suggested that bowing of poison strips might have led to bending of control rod guide shrouding, with a resultant narrowing of the control rod channels. This is no longer offered as an explanation of "rod stickings" experienced during reactor operation. In one instance (November, 1960), a "burr" was removed from one control rod guide shroud.[30] This did not eliminate stickings in this location. Indeed, after the burr removal, a total of nine stickings occurred within a period of $1\frac{1}{2}$ months, out of a total of 20 such occurrences between May, 1959 and the end of 1960.[31]

Post-incident examination of the shrouds and the control rod blades has revealed that only a thin film of corrosion was present. Rub marks and scratches found on the guide shrouds and control rod blades are attributed to normal wear during pre-incident operation. No evidence of inward bowing or warping of the shroud, or abnormal wear of rods or shrouding was found. Also, there was no evidence of a buildup of crud in the control rod channels.[10]

Among other possible causes of what has been termed "control rod sticking" are:

(1) Crud in the rod drive seal;

(2) Crud in, and wear of the Stellite, pinion-support ball bearings;

(3) Crud in, and wear and corrosion of the replacement pinion-support standard-alloy carbon steel ball bearings. It was found that, due to wear, the Stellite, pinion-support ball bearings were operating somewhat erratically. Subsequently, because Stellite bearings were not readily available, standard-alloy carbon steel bearings were used[32] and were replaced from time to time. These ball bearings are highly susceptible to corrosion.

(4) A "hydraulic piston" effect - a momentary partial vacuum in the space above the rack as the rod was dropped. This effect had been noticed during the testing of a prototype rod drive mechanism at Argonne National Laboratory. Consequently, the washer on top of the rack was modified (reduced slightly in diameter) to permit a more rapid equalization of pressures

between the reactor and the top of the rack housing. This modification appeared to have eliminated the effect, as shown by further testing. A piston effect of this sort has been observed on a similar drive mechanism, intended for another reactor.(3) The degree of this effect is influenced by such parameters as reactor pressure and water temperature, seal-water inlet temperature, and the level of water in the rack housing.

In summary, no one cause of the stickings has been established. It is very likely that there were several contributing factors.

VIII. CRITIQUE OF THE CONTROL SYSTEM; GENERALIZATIONS

A. Top Mounting vs Bottom Mounting of Control Rod Drives

Early in the post-incident investigation, several possibilities were discussed which illustrate the decreased accident potential of bottom-mounted control rod drives. It was noted earlier in this report that, for the ALPR, the top mounting of rod drives was essentially mandatory. A possible cause of the incident, suggested during the post-incident investigation, was that a gas explosion had occurred in the pressure vessel. When it was discovered that a nuclear transient had occurred, it was conjectured that hydrogen had accumulated in the vessel, had exploded, and had raised one or more control rods far enough to initiate a flux transient. In the SL-1, the likelihood of a hydrogen buildup was extremely remote, in the state of assembly existing at the time of the incident, because of the relatively large clearances in the drive mechanism components. However, in another reactor design with top-mounted drives, such an accumulation of an explosive gas might occur. If there were a gas explosion, it is conceivable that control rods would be raised and that a nuclear transient might result. In the case of bottom-mounted rod drives, usually the lower grid structure serves as a positive stop of the control rods in their position of full reactivity control. Thus, a pressure transient would tend to hold control rods in the configuration of maximum effectiveness.

In the SL-1 incident, control rod guide shrouds collapsed soon enough to trap the off-center control rods essentially in the position of full insertion. When the shield plugs were ejected from the vessel, the off-center rods were held by the shrouding, and breaks occurred at weaker points, in the drive mechanisms. Otherwise, withdrawal of the center control rod might have led to a raising of other control rods, and possibly to a more energetic transient.

For other reactor designs where control rod drives are to be mounted on the top head or bottom head of a pressure vessel, one should consider the following merits of bottom mounting:

(1) Bottom mounting of drives would reduce congestion of equipment on the top head of the pressure vessel, making access easier.

(2) Whenever the top head of the pressure vessel is to be removed, for example for fuel handling, the system of control rods could remain fully operative. Among the obvious advantages are:

(a) accurate indication of rod position at all times;

(b) the possibility of using rods as "cocked" safety rods; and

(c) less frequent handling of control rods.

In judging relative advantages of top mounting and bottom mounting of control rod drives in fluid-cooled reactors, there are disadvantages of bottom mounting that must be weighed. Among these are:

(1) In a compact plant, it might be difficult to provide the additional space required below the reactor vessel for the drives and for their maintenance. Also, additional shielding below the reactor vessel might be required.

(2) Corrosion products and other debris tend to collect or settle in drive thimbles at the bottom of the reactor vessel and interfere with drive mechanism operation. Since some of this material is radioactive, drive mechanism maintenance and replacement would become a more difficult task.

(3) Openings in the bottom head of the pressure vessel increase the likelihood of an inadvertent draining of coolant from the vessel. In the event of rupture of a drive thimble, it would be difficult to maintain adequate shutdown cooling of the fuel assemblies.

The choice between top mounting and bottom mounting of the drive mechanisms must be specific to the particular reactor design, although, in general, bottom mounting appears to be preferable.

B. Assembly (Disassembly) of the Drive to Its Control Rod

If the SL-1 incident occurred as postulated (Section III), three aspects of the control system design and assembly procedure were factors:

(1) The reactor could achieve super-prompt criticality, with the center rod partially withdrawn, even though the other four rods were inserted fully into the core.

(2) The procedure for attaching a drive mechanism to its control rod required that the rod be raised manually.

(3) Mechanical stops, which would limit the upward manual motion of the control rod, were not included.

A discussion of item (1) is presented in Section VIII-E. Items (2) and (3) are discussed below.

In the procedure of disassembling a drive from its control rod, in SL-1, once the seal and rack housings were removed, the control rod rested on the internal dashpot spring. The next step was to raise the control rod a short distance in order to attach a C-clamp to the rack so as to prevent it from dropping while the nut and washer were being removed from the top of the rack. In this step, the normal procedure required the cooperative efforts of two men, one to lift or hold the rod, and the other to attach or remove the C-clamp. Subsequently, this clamp was to be removed and the control rod lowered to its rest position.

In formulating the instructions for operations personnel for the disassembly (or assembly) of a drive mechanism from its control rod, account was taken of the hazard of withdrawing the center control rod too far. The instructions stated explicitly that no control rod was to be raised more than 4 in.[23] during the disassembly (or assembly) procedure. In other words, the control rod was not to be raised beyond an indicated position of 4 in. (Fig. 5). The normal procedure was to work with only one control rod unit at any given time. The other control rods were to be inserted fully. However, very little reactivity would have been added had all five cross control rods been at 4 in., indicated position, and the control margin was sufficient to override this extra reactivity. At this point, because the absorber overlapped the core when the rod was at indicated zero, the bottom of the cadmium section was approximately one (1) in. above the bottom of the active reactor core. At the time of the incident, with the other four cross-shaped control rods at indicated zero, it is estimated[9] that, to reach delayed criticality, the center rod would have had to be raised at least one (1) ft more than the instructions specified. This left a substantial safety margin for manual manipulation of the control rods.

Two other factors which lessened the likelihood of a fast rod withdrawal into the region of supercriticality were the weight of the mass being lifted (85 lb in water), and the congestion of equipment on the pressure vessel head, which tended to restrict bodily movements.

In spite of these various factors, acting to reduce the probability of a nuclear incident being caused by manual withdrawal of the center control rod, it appears that just such an incident, in fact, did occur.

It is now evident that the procedure selected to connect the control rod to its drive, though simple in concept, had an essential disadvantage, namely, the requirement of manual lifting of the rod. This disadvantage could not have been overcome by the addition of one or more mechanical stops, positioned so as to limit the upward travel of the rod during the assembly. It was not intended that the operation of attaching the drive to its control rod would be performed often, or that it would be a routine procedure. Nevertheless, less reliance should have been placed on the following of instructions and on the margins for error described above. In retrospect, it is clear that a different method of disassembly and assembly should have been designed, namely, one which did not require raising of the control rod from its position of full insertion. Reliance on mechanical stops is illusory, for, at some point during the procedure, the stops must be engaged and then they must be removed. Personnel performing the assembly operation might become so dependent on the safeguard afforded by these stops that a lessened vigilance could lead to an accident because of an unrealized failure to actuate them.

It must be remembered that, in typical control system designs, there is normally no indication of rod position when the control rod is not attached to its drive mechanism. This ignorance of rod position compounds the hazard of assembly, and it exists whether the control rod is moved manually or by other means. Therefore, such an assembly/disassembly operation requires careful supervision, and must not be considered routine. In this type of operation, as well as in others involving a possible increase in reactivity, use of a checklist system would tend to lessen the hazard.

C. <u>Burnable Poison</u>

In the early stages of the design of ALPR, various thermal-neutron absorbers were considered for use as burnable poisons.(33) Among those nuclides considered, having microscopic thermal neutron absorption cross sections in the range of hundreds of barns to thousands of barns, were Li^6, B^{10}, Eu^{151}, and Hg^{199}. These absorbers offer advantages of burning up slowly enough, in a thermal neutron flux of the order of 10^{13} n-cm/(cm^3-sec), to supply reactivity at a rate roughly commensurable with the rate of loss of reactivity resulting from the consumption of U^{235}. The B^{10} isotope was considered to be particularly useful for ALPR, for several reasons:

(1) the B^{10} is consumed at a sufficiently high rate that little of it would have been left, by the end of a core cycle, in regions of high statistical importance;

(2) there was more over-all experience with boron, as a burnable poison, than with any of the other isotopes mentioned above; and

(3) B^{10} was available, in a reasonably pure form of boron highly enriched in B^{10}, at a relatively low cost.

The basic ALPR core design involved the use of boron, highly enriched in B^{10}, as a constituent of the fuel-plate core. The intention was to provide a negative reactivity contribution of approximately 11%, in the cold fresh reactor. This was to counteract most of the positive reactivity effect of adding U^{235} to the fresh reactor to meet the requirements of power production for a core cycle of three years. It was not until contractual research and development programs failed to produce suitable fuel plates that the design concept of mixing fuel and burnable poison was altered.(3)

Much theoretical work has been performed to examine the subtleties of using burnable poisons in discrete masses, such as rods or strips. (For example, see Ref. 34.) Even when a poison supposedly is homogeneously distributed in a fuel-plate core, there may be a significant loss in effectiveness because of the microscopic lump structure.(35) Ideally, with a freedom of choice permitted in varying the location of the poison in the reactor core, and in varying the mass of poison per unit surface area of the poison lump, one can minimize the reactivity variation resulting from equilibrium power production. However, for single-batch core loadings, additional fissionable material would be needed initially to counteract the negative reactivity effect of the neutron poison remaining at the end of the core cycle. The amount of this extra fissionable material would depend upon the statistically averaged relative absorption rates of the burnable poison and the fuel, and upon the statistically averaged fractional burnup of fuel atoms.

Often, a designer attempting to use burnable poison in a power reactor operating at high temperature encounters problems of fabrication and/or problems peculiar to the particular choice of neutron poison or the form in which the poison is to be introduced. In the case of the burnable-poison strips of ALPR (Section IV-B), there had been earlier experience, in the BORAX-III core, with similar strips fabricated by powder metallurgy techniques.(14) This favorable experience of some six months use of burnable-poison strips in BORAX-III proved to be misleading, in terms of an expectation that strips of this type would be satisfactory for the anticipated core cycle of three years. Since the incident, additional experimentation at ANL has shown that poison strips as thin as the ones used in ALPR are not strong enough to resist the forces of a volume expansion resulting from the corrosion of the strip.(36) Probably, this growth in strip size in SL-1 caused a bowing of the strip in the regions between points of constraint where the strip was fusion-welded to a side plate of a fuel assembly. Even now, considering information available concerning the influence of neutron radiation on the corrosion rate of a poison strip of this type (Section VII-A), it is not known whether the use of much thicker strips, e.g., twice as thick as those used, would have ensured that the burnable-poison strips would remain in good condition for the entire core cycle. For that matter, it is not certain that, if the B^{10} had been dispersed in the core matrix, the fuel plates would have remained in such excellent condition for a three-year core cycle.

The unfavorable experience, in SL-1, with poison strips having a carrier of aluminum-nickel alloy should be instructive in two major respects. First, it is clear that extrapolation from an experience of a relatively short period of prototype testing to a situation where the item is to be in service for a very much longer period of time is uncertain, at best. In the event that such an extrapolation is forced upon the designer, a special effort should be made, during subsequent reactor operation, to check on the continued serviceability of the item at specified, regular intervals. This is not a novel suggestion, but all too often such checking is not performed often enough, or is performed haphazardly. Second, since burnable poison is an agent of reactor control, it is all the more important that it remain in its intended location and contribute a positive reactivity component at an expected rate. A dramatic deviation in the control requirement of the adjustable control system, either in the form of a sudden change or in the form of a sustained and steadily increasing deviation should be cause for a prompt, detailed examination of the poison carriers.

Burnable poisons can serve a variety of control purposes in reactors, and it is likely that they will receive increasing attention. In boiling-water reactors, another control scheme that could provide shim control and, possibly, essentially the entire control of the reactor, is that of soluble poison.

D. Soluble Poison

The use of soluble neutron poisons for control of thermal, water-moderated reactors has been discussed in several reports and articles.(37,38) At various stages in the formulation of the ALPR design, serious consideration was given to the use of boric acid solution for: (1) reactivity control at temperatures ranging from ambient to operating (420°F); or (2) routine, auxiliary shutdown of the reactor; or (3) non-routine, auxiliary shutdown. In the following paragraphs, attention is focused upon the problems, technical and/or logistical, which led to a decision to use boric acid only for non-routine, auxiliary shutdown of the ALPR.

It should be realized that special logistical needs played a vital role in this decision for the ALPR. Since then, soluble poison has been employed in other ways for control of boiling water reactors, as discussed briefly in the second part of this section.

The ultimate reactors, for which the ALPR was to serve as a prototype, were to be situated in geographically remote locations. In the design specifications(2) it was indicated that a source of fresh water might or might not be available, other than melted snow, and that this water probably would contain much mineral and organic matter. Also, in view of the difficulty and the cost of transporting plant components to these sites, considerable emphasis was placed on the design of a small, compact plant. Consequently, a basic criterion adopted by the reactor designers was the minimization of requirements of water supply, storage of demineralized water, and storage of radioactive waste water.

Since the reactor plants were to relieve problems of logistics in supplying petroleum for diesel generators, it was important to minimize the dependence on power from these generators which were to serve for emergency power production. For this reason, it was stipulated that "the nuclear plant must be capable of restarting after a shutdown occurring anytime during the core life" (Appendix A, Operational Requirements, 3). In particular, this meant that it was necessary to be able to take the reactor to full power after shutdown.

Auxiliary Shutdown of ALPR

If, as a routine procedure of reactor shutdown, enough boric acid were introduced into the reactor water to keep the cold, xenon-free reactor subcritical with any one control rod withdrawn, it is doubtful that the aforementioned criterion of power operation would have been met. A compromise would have been to add a smaller amount of boric acid, to reduce the magnitude of the excess reactivity potential in withdrawal of the center control rod.

Removal of the boric acid from the reactor water could have been accomplished by one of several methods, each method requiring additional equipment and/or additional material or storage space:

(1) by pumping reactor water through ion exchange columns containing a special resin, and returning the clean water to the reactor;

(2) by distilling reactor water containing boric acid, and returning the clean water to the reactor; or

(3) by massive dilutions of the reactor water with demineralized water.

In principle, methods (1) and (2) above, are quite similar. In both cases, the initial concentration, $N(0)$, of boric acid in the reactor water is reduced according to an exponential law:

$$N(t) = N(0) \exp(-t/T) \quad .$$

Here, N(t) is the boric acid concentration at a time t after the initiation of cleanup; T is the time required to process, on a once-through basis, a volume of water equal to that in the pressure vessel, the purification system, and any other components which contain water to be purified; and the assumptions have been made that cleanup is 100% effective, and that the clean water is mixed uniformly and continuously with the water being purged. Typically, in a boiling water reactor, the coolant fluid is kept clean by low-rate, continuous purge. This was the case for the ALPR. Consequently, unless special provisions had been made to increase substantially the rate of cleanup, for example, by use of a separate purge system, it would have taken many hours to remove, say, 90% of the boric acid.

If ion-exchange columns had been used for removal of boric acid, a large inventory of costly resin would have been required, sufficient for the period between times of accessibility for supply to the remote site. Cleanup by distillation would have required additional equipment, including means of flushing the residue, and probably would have involved extensive use of diesel-generator power if full auxiliary shutdown control by boric acid were employed.

The third method, that of massive dilution of the reactor water, would have required a much larger supply of demineralized water and a larger sump for radioactive waste water.

The reactor designers concluded that the special logistics problems of the ultimate plant made it highly undesirable to use soluble poison for routine auxiliary control for shutdown. Therefore, the soluble-poison system of ALPR was designed for non-routine, backup control, to be used at the discretion of the operations supervisor.

In the prototype plant itself, there was less reason not to add boric acid solution routinely in the process of reactor shutdown. Cleanup by ion exchange was feasible, though costly and slow. Moreover, this procedure of shutdown would have been special to the prototype plant. Nevertheless, if such an incident as did occur had not been considered to be so unlikely, probably the routine addition of boric acid would have been made an integral part of the procedure of shutdown of the prototype reactor whenever assembly of the drive to (or disassembly of it from) the central control rod was to be performed.

Operational Control with Soluble Poison

For an early conceptual version of the ALPR, consideration was given to a control system where boric acid, in solution, would supplant the customary control by rods, with the exception of perhaps one control rod available and cocked as a safety rod.(39)

It is doubtful that a good core design would have permitted the use of enough soluble poison to preclude the need for an auxiliary control in the form of a burnable poison. Without burnable poison, so much control capacity would have been needed that there might have been a large initial positive void coefficient in the cold fresh reactor. Also, at this time, not enough data had been accumulated from operating boiling-water reactors to support the use of a soluble-poison control system at operating temperature (420°F) in the ALPR.

Now, there is more information available about the use of soluble poison for control of boiling water reactors. Boric acid has been used for shim control of the YANKEE reactor.(40) In the course of the recent experiments where the EBWR power level was raised to 100 Mwt, boric acid was used to assist control of the reactor. The H_3BO_3 concentration was adjusted as reactor power was varied. These are rather special uses of soluble poison, and they do not adequately demonstrate the full utility of soluble poison in boiling water reactors, namely, for control of a power reactor at operating conditions for long periods of time. There remain questions as to how well a particular boiling-water reactor would perform with a dilution/concentration procedure of routine reactor operation with soluble-poison control. A reliable procedure for controlling the boric acid concentration in the reactor water would be required. Also, it would be useful, but perhaps not essential, to have a procedure of sampling the reactor water at regular intervals, or continuously, and determining the boric acid concentration. A typical cleanup procedure (e.g., ion exchange) is slow. On the other hand, it is precisely this slowness of cleanup that is one of the safety advantages of control by soluble poison, for unintentional removal of poison from the reactor water would be a slow process.

E. Reactor Shutdown Margin

The concept of "reactor shutdown margin" has been defined rather poorly, in the past, in terms of its significance for an evaluation of hazards associated with a given reactor design. Sometimes, what is listed as the shutdown margin is the numerical value of the shutdown margin in the fresh reactor. This is defined to be the per cent reactivity by which the reactor is subcritical, with the normal control system fully effective, and with the reactor otherwise at its maximum reactivity. Typically, the reactor possesses maximum reactivity when it is cold and there is minimum void and xenon in the reactor core. This concept of shutdown margin is too limited, for often the fresh reactor offers less of a hazard than a reactor which has been operated at power and in which fission products have accumulated.

More generally, the shutdown margin is the smallest value of reactivity that must be added to bring the reactor to delayed criticality, at any time in reactor life, and under all possible reactor states of temperature, pressure, etcetera, when the normal, adjustable reactivity-control system is exercising its maximum effect. If, as in the case of the SL-1 reactor, a backup, non-emergency control system is available, some of this auxiliary control may be added when needed to increase the shutdown margin. However, unless such auxiliary control is introduced automatically, and rapidly, its effect should not be included in the determination of the numerical value of the shutdown margin.

It has been pointed out(41) that even the more general concept of shutdown margin may be inadequate as a criterion to be used in assessing the safety of a proposed reactor. Bates(41) has assembled data concerning the shutdown margin for a number of power reactors, illustrating a remarkably large variation in what different design groups have considered to be acceptable or desirable. One should not infer that a given reactor design, with an actual shutdown margin of, say, 6% reactivity, would have been unacceptable if the margin had been only 4%. Normal design practice requires that the calculated shutdown margin be conservative, relative to actual needs, to allow for uncertainty in the calculation.

A margin of reactor control is required for a variety of reasons, and the size of this margin is set, in part, by these reasons. For example, the shutdown margin should be large enough so that one need not be overly much concerned about the reactivity hazards of small errors in fuel loading. In the SL-1 reactor, the shutdown margin was in excess of 2% reactivity, and probably at least 3%, even without the total of six cadmium blades added (November, 1960) in two of the four tee-rod locations for additional fixed control. (The cold fresh reactor was subcritical by approximately $3\frac{1}{2}$ % with all five cross-shaped control rods at indicated zero.) This margin was more than enough to override such possible mistakes in fuel loading changes as:

(1) the inadvertent loading of a fuel assembly without poison strips;

(2) a misorientation of a fuel assembly; or

(3) the addition of an extra fuel assembly at the periphery of the reactor core.

Moreover, it appears that, at any time during core life, removal of any one of the off-center cross-shaped control rods would have left the reactor subcritical if the four remaining cross-shaped control rods were inserted fully. Yet, the shutdown margin was insufficient, in the sense that presumably it was the withdrawal of the central control rod that led to the transient that destroyed the SL-1 core on January 3, 1961.

F. The "Single-Mistake" Criterion

In the case of SL-1, each cross control rod was of a large span (14 in. of absorber), and the rods were so far apart that withdrawal of the center rod effectively left unprotected a portion of the core that was large in terms of the characteristic diffusion distance of thermal neutrons. As a result, the cold fresh reactor was supercritical in the absence of the center control rod. So far as can be ascertained at this time, the SL-1 incident could have been brought about, in the same manner, in the fresh reactor.

It would be tempting to state categorically that no future reactor design should be accepted if, at some time in reactor life, and at certain conditions of temperature, pressure, etcetera, withdrawal of a single component of the control system (e.g., a single control rod) would result in reactor supercriticality even though the rest of the control system was fully effective. Certainly, it should be a prime goal of the design effort to avoid such a possibility. It may be, however, that in the avoidance of this undesirable situation, the reactor design would be affected adversely in other important respects. In such a case, the reactor designers must arrive at a decision as to the extent of a compromise of the conflicting goals.

In the ALPR, the gearings of the rack-and-pinion drive were chosen so as to limit the reactivity insertion rate to approximately 0.01%/sec in the cold fresh reactor.[42] This rate is in sharp contrast to the very much higher reactivity insertion rate that could be achieved by manual rod withdrawal, when the drive was disconnected from its rod. From data obtained and tests performed by Combustion Engineering, Inc.[7] and by the General Electric Co.,[9] it may be inferred that a reactivity addition rate of perhaps 20%/sec could have been obtained by a rapid manual withdrawal of the center control rod at the time of the SL-1 incident. This result dramatically illustrated a possible hazard of rod handling in a shutdown reactor, whether or not the SL-1 incident was caused in this way.

In some reactor designs it may be feasible to require that, before any control rods may be handled manually, enough fuel must be removed from the shutdown reactor so that it would remain subcritical even if all control rods were withdrawn fully. The fuel would be stored in a suitably controlled facility until the control rod drives were reassembled to their rods.

One alternative to this procedure is to introduce temporary additional control as a <u>routine</u> procedure in the process of reactor shutdown. For example, in the case of a water-moderated reactor controlled by a system of control rods, perhaps boric acid solution could be introduced into the reactor water during shutdown, for additional protection of the cold reactor. Soluble poison was not used in this manner in ALPR, for reasons stated in Section VIII-D.

A less restrictive condition that may be acceptable is that no single mistake in manipulation, and no single malfunction of one component of the reactivity-control system should lead to an excursion that would damage the reactor or result in a serious hazard to personnel.

Thus, for example, if, upon full withdrawal of a single control rod, the system would be below prompt critical, at worst a relatively slow rise in the level of neutron flux would result, and the reactor should be capable of compensating automatically for this excess reactivity, either because of negative reactivity coefficients or by virtue of an auxiliary and/or emergency control system, without a transient that would damage the reactor or would expose personnel to a severe radiation hazard or to other serious injury. In any event, the reactor design should be such that, whether or not the reactor is damaged, personnel in the area would not suffer serious injury; for some types of reactors, this less restrictive criterion might be satisfied even if the net uncompensated reactivity was more than one (1) dollar.

IX. SUMMARY AND REMARKS

The SL-1 incident has focused attention on certain design features and operating procedures of the prototype nuclear reactor power plant. In this report, emphasis has been placed on the design and the performance of the ALPR (SL-1) reactivity control system. Criteria for the reactor and the attendant motivations for the control system design have been reviewed.

The post-incident investigations and reviews have spotlighted such diverse features of reactor design and performance as: deterioration of burnable-poison strips; top mounting of the control rod drives; "sticking of control rods"; the fact that at least partial insertion of the center control rod was required for reactor shutdown under certain conditions; and the procedure of assembling the control rod drives to their respective control rods.

It is now clear that such an incident, in principle, could have occurred in the unoperated prototype reactor, under conditions where the burnable-poison strips were intact and where the control rods operated freely.

Under certain circumstances, in other reactor designs involving the use of top-mounted control rod drives, a non-nuclear accident might initiate a nuclear excursion. For example, a gas explosion inside a pressure vessel would exert a force on components of the control rod drives, possibly lifting the control rods. It has been shown that such an event did not occur in the SL-1. Thus, top mounting played only an indirect role in initiating the incident, largely through its influence on the procedure of attaching drives to the control rods, in terms of the sequence of events postulated for the incident.

Various designs of systems of control rods were considered for the ALPR, in an attempt to secure a system where reactor shutdown was assured in the event of a full withdrawal of any one control rod. The final choice was a design compromise wherein the control provided by the center control rod plus any three of the four other control rods was sufficient for shutdown of the reactor in its state of highest reactivity. The designs of the control rod and drive were tailored to a simple assembly-disassembly procedure, which involved a manual lifting of the control rod to a position only a few inches above its position of full insertion. Written instructions for the procedure of assembling a drive to its control rod specified that the rod was not to be raised more than 4 inches.(23) This left a margin of at least one (1) ft additional withdrawal of the center control rod, even to the position of <u>delayed</u> criticality. Other features of the reactor design militated against a withdrawal of a control rod so far and so rapidly as to cause a dangerous nuclear excursion. Yet, all available evidence indicates that the SL-1 excursion was caused by a manual withdrawal of the center control rod.

In retrospect, it is clear that less reliance should have been placed on awareness of hazards and on written instructions. Perhaps a checklist procedure would have prevented this particular incident, but this falls into the same category. Instead, a different assembly procedure, not requiring the manual raising of control rods, should have been designed. Also, there should have been even greater emphasis on achieving a reactor design wherein, when the other control rods were inserted fully, rapid withdrawal of a single control rod could not lead to a catastrophic reactor excursion.

One must remember that, so long as a reactor can be made supercritical by the motion of a control rod, it is conceivable that a nuclear incident could be initiated by rod withdrawal. This is less likely if the reactor can be shut down with any single control rod withdrawn. However, the satisfaction of the latter condition does not eliminate the possibility of such an event. For example, if some of the other control rods were withdrawn partially, a fast partial withdrawal of one control rod then might lead to a short-period excursion.

Reactor designers must take into account the practical aspects of actual reactor operation. They should communicate to the reactor operators their motivations for the particular choices of reactor design, stressing known potential hazards of the design selected. It would be useful if the designers also indicated disadvantages and potential hazards of alternative, rejected designs, for then the reactor operators would be made more cognizant of the problems which they might encounter because of seemingly innocuous modifications of reactor design or operating procedure. The reactor operators should consider the significance to reactor safety of an accumulation of small modifications, periodically reviewing the state of the reactor, preferably with an independent evaluation by a competent group not associated with the operation of that particular reactor.

In the final analysis, vigilance and deliberateness of action on the part of the reactor operators must be stressed. All operating personnel should be made fully aware of known potential hazards of any of the operations which they might be called upon to perform, and the consequences of a violation of instructions. No task involving a possible gain in reactivity should ever be considered to be routine. The written, detailed instructions for these operations must be clear and unambiguous, and the existence of these potential hazards should be stated in boldly evident precautionary notes.

APPENDIX A*

TECHNICAL CHARACTERISTICS
ARGONNE LOW POWER REACTOR (ALPR)
(March 9, 1956)

Note: The following characteristics include Change No. 1 (May 1, 1956), Change No. 2 (June 14, 1956), and interpretation of the specifications ensuing from a meeting between Army Reactors Branch and Argonne National Laboratory personnel on August 21, 1956.

The Argonne Low Power Reactor Project (ALPR) has been established by the Atomic Energy Commission, at the request of the Department of Defense, as an ultimate member of a family of nuclear power plants to produce electric power and heat at remote military bases.

The ALPR should be designed to provide the best possible boiling heterogeneous nuclear power plant system that meets the requirements of an Auxiliary Station, DEW Line, and is compatible with arctic environmental conditions. The plant will include the structure housing and auxiliary systems. The prototype plant will be constructed at the National Reactor Testing Station, Idaho. The purpose of prototype construction is to obtain information and data on new features of the system. A major objective of the prototype design is to minimize the engineering modifications that need be made when adapting this design to a specific Auxiliary Station.

Construction of the prototype power plant system should include as many features peculiar to arctic conditions as is possible. For example, the air space required between grade and the bottom of the plant structure will be provided. Whether such air space is provided in the prototype structure by placing it on piling, as it must be in the arctic, or by some other construction design, may be determined by considerations of local economy.

Whenever it is necessary to deviate from arctic practices, it is desired that appropriate recommendations be made through the Chicago Operations Office to the Army Reactors Branch, Division of Reactor Development, Washington, D. C.

These Technical Characterisitcs are to be construed as design objectives for the prototype plant and not as performance guarantees. Changes may be made to ensure the design of a simple, reliable nuclear power plant

*Reproduced from Appendix V of Ref. 1: ANL-6076, "Design of the Argonne Low Power Reactor (ALPR)," with references deleted.

for the intended application. Recommendation for changes or modifications should be made to the Army Reactors Branch through the Chicago Operations Office.

The DEW Line is now under construction. Specific information on loads, load factors, and curves are not available. A specific site for the installation of the first operating unit has not been selected. Therefore, for many items in this specification, only "average arctic conditions" are given. As these and other application information becomes available, data will be furnished to Argonne National Laboratory.

It is desired that, insofar as possible, the materials, components, and systems of the nuclear power plant, especially for the reactor component, should be acceptance tested under simulated operation conditions before prototype construction.

Washington supervision of this project rests in the Army Reactors Branch, Division of Reactor Development, U. S. Atomic Energy Commission.

General Requirements

1. Capacity (Design)

   200 kw Electrical
   400 kw Heat, Net (approximately 1,300,000 Btu/hr)

2. Frequency

   60 cps

3. Voltage (Busbar)

   120/208, 3-phase, 4-wire, wye connected

4. Power Factor

   0.8

5. Plant Factor

   0.7

6. Standby Equipment

   Full capacity, electrical and heating, in conventional diesel-electric power plants and/or oil-fired furnaces assumed.*

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*The full-capacity electrical requirement is for an arctic installation. The prototype installation was approved with 60-kw diesel-electric.

7. Transportability

Sealift, airlift, and overland transportation are available in the Arctic. Sealift is used only during the summer months, whereas airlift may operate the year round. The runways at some DEW Line sites become soft during the summer which prevents aircraft operation.

Aircraft operating to DEW Line stations are C-123 and C-124. Details on cargo preparation for these aircraft will be furnished separately.* The prototype plant must be designed to incorporate this air transportability feature, by components, prior to initial operation.

8. Personnel

Every man stationed at a remote arctic site increases the logistic support problem for that location. For this reason, the number of operating personnel to run this plant must be a minimum. It may be assumed that certain members of the organization will have some training in the operation of this plant to the extent of routine operation, inspections, and preventive maintenance. Such training will include basic steam technology, power plant operation, parallel operation of generators and instruction in reactor operation; health physics, nuclear instrumentation, etc. These individuals undoubtedly will be utilized in the discharge of the primary mission of the remote site.

The supervisory personnel at this remote site may be expected to have very little knowledge of reactor technology. The reliance for uninterrupted operation is placed upon the technological development of the plant components, the plant design and the operating personnel. Thus, a nuclear plant that will operate reliably for prolonged periods of time with the minimum of supervision and logistic support is required.

The military personnel assigned to remote arctic stations are relieved every 12 months, or less, and replaced by a new group. It should be assumed, therefore, the background and training of the incoming personnel is the bare minimum to satisfactorily accomplish their mission.

The present plans are to operate the DEW Line by contract. It should be assumed the initial operators will be civilians. Major maintenance, fuel changing and radioactive waste disposal will be accomplished by especially trained personnel, civilian and/or military.

*The air transportability feature limited all "packages" to $7\frac{1}{2}$ ft x 9 ft x 20 ft, and 20,000 lb in weight.

9. Facilities

The buildings designed for the Auxiliary Station of the DEW Line consist of modules* approximately 16 ft x 28 ft x 16 ft. These modules are connected in one line so that the ultimate structure is 28 ft wide and up to 412 ft long, depending on the number of modules used. The radar antenna dome is above these modules. Each module, the assembly, and the radome is designed for winds of 125 mph, 2 in. of ice, and 30 lb/ft^2 of snow.

The modules are designed to be compatible with arctic conditions:

(a) Comfortable, adequate, and flexible to meet personnel and equipment requirements.

(b) Resistant to fire, wind, cold, storm, and deterioration.

(c) Simple and economical to transport, construct, and maintain at arctic sites.

The building design is to emphasize maximum resistance to, prevention of, detection of, and control of fire, consistent with associated problems of arctic construction, maintenance, operations, economics, and logistics.

The floor of the antenna in the radome over the building train is 50 ft above the local terrain. The highest point of the building housing this power plant should not be higher than 50 ft above ground. Should this height be exceeded, the Army Reactors Branch, AEC, should be so informed before the design is approved.

Outside access doors must not be encroached upon by building design lest the requisite area for snow removal by mechanized equipment be hampered.

The building housing this plant should conform to these general DEW Line requirements.

Operational Requirements

1. Environmental Conditions

Outside ambient temperature: -60°F to +60°F. The plant should be designed for 60 kw electrical overload at 40°F outside ambient air temperature. (See No. 4 below.)

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*The module requirement is for an arctic installation. The prefabricated support facilities structure for the prototype was approved.

Winds: The specifications on the present structures require a building to withstand 125-mph gusts, 30 lb/ft^2 of snow, and 2 in. of ice.

Permafrost: Permafrost may be expected throughout the DEW Line region. Thawing of the permafrost introduces undesirable stresses in the buildings. Adequate protection must be provided to preserve the permafrost regime. Should refrigeration be required, this electrical load must be considered parasitic.

Annual precipitation: The arctic coast, in general, has very little precipitation. That section of the coast in Alaska proposed for the first installation receives approximately 4.5 in./yr.

Water availability: Many of the DEW Stations have available a source of fresh water, other than melted snow. This water may be expected to have a high mineral and organic material content. Detailed information will be made available at a later date.

Site materials: Information on this subject is limited. Gravel has been used at many DEW Line Stations. Detailed information will be furnished at a later date.*

2. Routine Operation

Upon successful completion of acceptance tests, the operational nuclear plant is expected to be placed in routine operation. The plant will be prepared for operation by an operator in accordance with standard procedures for starting up the plant. When ready to take the electric and/or heating loads, parallel operation with the conventional systems will be established, the load shifted to the nuclear plant, the conventional system shut down as prescribed by standard operating procedures. The plant will be adjusted to the load conditions and placed into "demand controlled operation" by the operator. When the plant is operating stably and to the satisfaction of the operator, further detailed and constant attention by operating personnel should be unnecessary. The operator will be performing other duties elsewhere. Routine inspections and preventive maintenance (such as keeping oil reservoirs filled) will be necessary.

Guidance on frequency and extent of inspections and maintenance is expected of Argonne National Laboratory, and the Architect-Engineer.

Whenever required and foreseen, the shutdown of the nuclear plant will be accomplished manually in accordance with standard operating procedures.

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*Gravel from a DEW Line Station was examined and appraised as being acceptable for biological shield material.

3. Restart

Normal startups will be under the control of an operator in accordance with standard procedures. This action will follow core replacements, scheduled shutdowns, and emergency shutdowns. The nuclear plant must be capable of restarting after a shutdown occurring anytime during the core life. Plant down time only reintroduces the petroleum logistics problem that nuclear plants are to relieve, hence, the need to restart at the earliest possible moment.

4. Plant Overload

This plant must be capable of a 60-kw (electrical) overload for a period of one (1) hour or less in each 24-hour cycle. Such a condition may result from the simultaneous utilization of equipment with low diversity factor or the energizing of radome heaters. Under this condition of operation, the plant must be stable and respond readily and automatically to demand control signals; an operator must not be required.

This overload requirement may be relaxed if a non-standard turbine-generator size should be necessary.

When in this overload condition, the reactor must have sufficient control capability so that for steady operation, or for sudden reduction of load, the reactor will not present any unusual hazard.

5. Stability and Regulation

The plant must be stable throughout its operating range, which is defined as any heating load up to 400 kw plus a net electrical load from 20 to 260 kw. The maximum load of 260 kw includes a 60-kw overload. The following voltage and frequency conditions must be observed:

(a) Voltage may not vary more than ±5% from the design rated value with load varying from 20 to 260 kw. Loads may be applied in increments of as much as 60 kw.

(b) Frequency must not vary more than ±5% from the design rated value with load varying from 20 to 260 kw.

(c) Voltage may not vary more than ±2% from the steady-state value at any given load within the operating range.

(d) Frequency may not vary more than ± 0.1 cps from the steady-state value at any given load within the operating range.

(e) Transient voltages may not vary more than ±5% from steady-state values. Steady-state voltage must be re-established within 5 sec.

(f) Transient frequencies may not vary more than ±0.5 cps from steady-state values. Steady-state frequency must be re-established within 5 sec.

If a bypass steam system is utilized to meet the requirement for control and 60-kw load increments, up to approximately 1300 lb/hr of steam may be bypassed from the main steam line through a heat exchanger to the condenser. A dead band of approximately 200-300 lb/hr will be allowed to avoid excessive operation of the control system. It must be possible to adjust manually the amount of steam which is bypassed. The bypass steam flow rate must be re-established within 30 sec.

The voltage and frequency requirements listed above are dictated by the requirements of the radar-communications equipment at the DEW Line Station. The sensitivity of this equipment to frequency and voltage changes is being investigated and as detailed data becomes available this information will be furnished Argonne National Laboratory. In order to provide this high-quality electrical power to the electronics equipment, the conventional plant is so arranged that transient-producing utility type equipment is isolated from the radar by using (a) separate diesel-generator(s). Consideration should be given to the isolation of the radar-communications equipment load by some suitable arrangement. However, if this is not feasible, this regulation of voltage and frequency must be applied to the full plant load. If the sensitive electronics load can be effectively isolated, the voltage and frequency regulation for the balance of the station load need only conform to current practice of ±5% voltage and ±2% frequency variations. The characteristics of the nuclear plant must be such as to provide for stable parallel operation under all load conditions with the conventional diesel-electric equipment now used.

6. Instrumentation

Instrumentation of the nuclear plant should be a minimum consistent with the safety of the plant and personnel. The prototype of this plant may be instrumented as necessary to meet the AEC Advisory Committee on Reactor Safeguards (ACRS) requirements, but analysis should be made and experimentation should be conducted on the prototype to confirm the minimum requisite instrumentation. Provision should be made for audible warning when hazardous operating conditions are imminent.

Unless otherwise available, controls and instrumentation sufficient for satisfactory parallel operation with the conventional plant must be provided.

The complete control operation - startup, manual and/or automatic load changes, parallel operation and shutdown - should be performed from one control panel. All visual and recording instruments of the nuclear plant must be available at one location. Duplicate instruments may be provided at desired locations.

Power Plant Components

1. Reactor

The reactor of the boiling heterogeneous type is to have the inherent ability to adjust the power generation, and hence, the steam output, to meet the load demand placed upon the power plant. The load demand may be heating or electrical and any combination of the two. The reactor must be stable at all reactor power levels from a "zero power critical" condition through a power level equivalent to a plant electrical overload of 50% throughout the core life. Provisions must be incorporated in the design to provide a sub-critical core at all times during the core or fuel assembly replacement operation and the subsequent cooling period.

Minimum mechanical equipment should be within the biological shield of the reactor. Components with rotating or movable parts should be readily available for inspection and maintenance.

Unless the reactor design is adequate to accommodate the dissipation of heat after shutdown, cooling must be provided, including the cooling of spent cores in storage, if necessary.

2. Component Contamination

The conceptual design places the turbine in the primary loop and outside of the biological shield. This planning requires maximum water purity, minimum entrainment of radioactive particles in the steam, high fuel plate integrity, and a system for the detection of excessive radioactivity in the primary loop.

3. Core Life

The life of the reactor core must be at least two (2) years at full design load. It is expected that fuel changes will be made every three (3) years. The fuel plates must withstand the effects of burnup, corrosion, erosion, heat generation, and retain the fission products for these periods.*

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*Since there was insufficient operating experience with aluminum-type X8001-clad fuel plates in a boiling water reactor, a firm guarantee of a 3-year core life could not be provided. The Army Reactors Branch was made cognizant of this fact early in the design phases of the program. An effort has been made to lengthen the operational life of the fuel plate by increasing the cladding thickness, and by using Al-Ni-U fuel plate "meat" that is almost as corrosion resistant in boiling water as the cladding material. It is believed that at the present (February 5, 1959) operating conditions of the reactor, the assemblies of fuel plates should fulfill the design goal of core life.

This requirement is necessary to hold down the logistic support of this plant and to minimize the time required to reach the point of more economic operation than by conventional plants. The plant down time for any reason must be a minimum in order to keep the petroleum consumption low.

4. Shielding

The shielding of the reactor component represents the greatest single volume of the plant, and emphasis should be placed upon realizing the smallest volume and weight of shield necessary.*

On the other hand, personnel at the remote site must not be exposed to a radiation dose in excess of 50 mr/7-day week. In the shield design, consideration may be given to the use of exclusion areas and/or shadow shielding, provided the above maximum dose rate is not exceeded. The personnel safety criteria for shielding the reactor apply to the shielding of the spent fuel storage pit.

5. Fuel Storage

Fuel storage for spent fuel assemblies of one core must be provided. The time spent in "cooling" is subject to economic evaluation. It may be assumed that a spent core will cool for one year, and possibly for one core life. Adequate safety provisions must be incorporated to maintain the spent fuel in a subcritical condition at all times during storage.

The fuel storage chamber should be so designed and situated with respect to the reactor and the power plant that removal of the spent assemblies is facilitated and the "cooling off" does not interfere with the operation and maintenance of the plant.

*The shielding criteria were based upon the necessity to transport all shield materials by air. The use of locally available gravel for shielding relaxed the requirement.

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